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Docket Number 50-346

License Number NPF-3

Serial Number 2325

September 29, 1995

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: License Amendment Application to Revise Technical Specifications and Associated Bases for Pressurizer Code Safety Valve Lift Setpoint Tolerances

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License NPF-3, Appendix A, Technical Specifications, to reflect the changes attached. The proposed changes involve Technical Specification (TS) 3/4.4.3, Safety Valves and Pilot Operated Relief Valve-Operating, and associated Bases 3/4.4.2 and 3/4.4.3, Safety Valves.

The proposed change to Technical Specification Limiting Condition for Operation (LCO) 3.4.3 and associated Bases modifies the lift setting of the pressurizer code safety valves (PSVs) to ≤2575 psig, which corresponds to a lift setting tolerance of +3% of the nominal lift pressure. The present Technical Specification specifies a lift setting of ≤2525 psig corresponding to a lift setting tolerance of +1% of the nominal lift pressure. Increasing the upper bound of the lift setting tolerance of the PSVs from +1% to +3% will allow normal surveillance testing of the PSVs to be within +3% of the nominal lift setpoint of 2500 psig and still be acceptable under the LCO. This change will still maintain a significant margin of safety. The +3% tolerance is consistent with ANSI/ASME OM-1, 1981 Edition, which specifies a ±3% as-found testing tolerance for safety and relief valves within the scope of the Inservice Testing (IST) program.

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In order to support the next scheduled tests of the Pressurizer Code Safety Valves, Toledo Edison requests that this amendment be issued by the NRC by July 1, 1996.

Should you have any questions or require additional information, please contact Mr. William T. O'Connor, Manager - Regulatory Affairs, at (419) 249-2366.

Very truly yours,

NKP/eld

cc: L. L. Gundrum, DB-1 NRC NRR Project Manager
H. J. Miller, Regional Administrator, NRC Region III
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Utility Radiological Safety Board

APPLICATION FOR AMENDMENT
TO
FACILITY OPERATING LICENSE NPF-3
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1

Attached are requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial Number 2325) concern:

Appendix A, Technical Specification 3/4.4.3, Safety Valves and Pilot Operated Relief Valve - Operating

Appendix A, Technical Specification Bases 3/4.4.2 and 3/4.4.3, Safety Valves

By: J. P. Stetz
J. P. Stetz, Vice President - Nuclear

Sworn and subscribed before me this 29th day of September, 1995.

Evelyn L. Dress
Notary Public, State of Ohio

EVELYN L. DRESS
Notary Public, State of Ohio
My Commission Expires 7/28/99

The following information is provided to support issuance of the requested changes to Davis-Besse Nuclear Power Station, Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specification (TS) 3/4.4.3, Safety Valves and Pilot Operated Relief Valve - Operating, and associated Bases 3/4.4.2 and 3/4.4.3, Safety Valves.

A. Time Required to Implement: This change is to be implemented within 90 days after NRC issuance of the License Amendment.

B. Reason for Change (License Amendment Request Number 94-0010, Revision 0):

The proposed change to Technical Specification Limiting Condition for Operation (LCO) 3.4.3 and associated Bases modifies the lift setting of the pressurizer code safety valves (PSVs) to ≤2575 psig, which corresponds to a lift setting tolerance of +3% of the nominal lift pressure. The present Technical Specification specifies a lift setting of ≤2525 psig corresponding to a lift setting tolerance of +1% of the nominal lift pressure. Increasing the upper bound of the lift setting tolerance of the PSVs from +1% to +3% will allow normal surveillance testing of the PSVs to be within +3% of the nominal lift setpoint of 2500 psig under the LCO. This change will still maintain a significant margin of safety. The +3% tolerance is consistent with ANSI/ASME OM-1, 1981 Edition, which specifies a ±3% testing tolerance for safety and relief valves within the scope of the Inservice Testing (IST) program.

C. Safety Assessment and Significant Hazards Consideration: See Attachment

Docket Number 50-346
License Number NPF-3
Serial Number 2325
Attachment

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 94-0010

(11 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 94-0010

TITLE:

Revision of the Davis-Besse Nuclear Power Station (DBNPS), Unit 1 Operating License, Appendix A, Technical Specification (TS) 3/4.4.3, Safety Valves and Pilot Operated Relief Valve - Operating and Bases Section 3/4.4.2 and 3/4.4.3, Safety Valves.

DESCRIPTION:

The proposed change to the DBNPS Technical Specification Limiting Condition for Operation (LCO) 3.4.3 and associated Bases modifies the lift setting of the pressurizer code safety valves (PSVs) to ≤2575 psig, which corresponds to a lift setting tolerance of +3% of the nominal lift pressure. The present Technical Specification specifies a lift setting of ≤2525 psig corresponding to a lift setting tolerance of +1% of the nominal lift pressure. Increasing the upper bound of the lift setting tolerance of the PSVs from +1% to +3% will allow normal surveillance testing of the PSVs to be within +3% of the nominal lift setpoint of 2500 psig and still be acceptable under the LCO. This change will still maintain a significant margin of safety. The +3% tolerance is consistent with ANSI/ASME OM-1, 1981 Edition, which specifies a ±3% as-found testing tolerance for safety and relief valves within the scope of the Inservice Testing (IST) program.

If the valves are found outside of a ±1% tolerance during surveillance testing, the setting will be adjusted to within ±1% of the specified lift setting prior to return to service. This is consistent with the ASME Code, Section III, 1971 Edition, Subarticle NB-7614.3, which specifies an initial setting tolerance not exceeding ±1%.

The most limiting overpressure transient for the PSVs is the control rod withdrawal event from a low power condition. This event has been reanalyzed and found to be acceptable with greater than a +3% lift tolerance on the PSVs (Reference 1). The present Technical Specification does not specify a lower bound for a lift setting tolerance and no safety analyses are affected by not specifying the lower lift tolerance.

Bases Section 3/4.4.2 and 3/4.4.3 will be revised to reflect the results of the new analysis, and to require the as-found lift tolerance of ±3% and the as-left lift tolerance of ±1%, consistent with the ASME Code.

These changes are shown in the attached, marked-up pages of the Technical Specifications.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

Reactor Coolant System
Pressurizer Code Safety Valve Lift Setpoints

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS AND ACTIVITIES:

The PSVs protect the Reactor Coolant Pressure Boundary from overpressure transients. In particular, by the ASME Code, Section III, 1971 Edition, Subarticle NB-7511, the peak RCS pressure is limited by the pressure-relieving devices to less than 10% above design pressure, or 2750 psig.

EFFECTS ON SAFETY:

The only effect on safety of this change is with regard to the +3% lift tolerance. The -3% lift tolerance has no effect on the safety analyses.

The control rod withdrawal accident from low power (startup accident) is the event which most significantly challenges the PSVs. The current calculated peak pressure for this event is 2713.4 psia (USAR Section 15.2.1). The peak pressure of the startup accident exceeds other overpressure events such as the loss of main feedwater (USAR Section 15.2.8) and the turbine trip followed by the high RCS pressure trip (Reference 2).

Generally, the startup accident models the control rod withdrawal as a reactivity insertion. At slow reactivity insertion rates (RIRs), the reactor trips on high RCS pressure. At more rapid RIRs, the reactor trips on high flux prior to the high pressure trip. The analyses were performed with a spectrum of RIRs to determine the RIR that results in the highest peak RCS pressure.

USAR Section 15.2.1 provides the following acceptance criteria for the startup accident:

1. The RCS pressure will not at any time exceed 110% of design pressure (2500 psig), or 2750 psig.

2. The reactor thermal power shall not exceed 112% of rated power.

To support this License Amendment Request, the present USAR startup accident analysis was reperformed by Babcock and Wilcox (B&W) to generically support increasing the lift tolerance on the PSVs to +3% for several B&W plants (Reference 1). The new analysis was performed using the RELAP5/MOD2-B&W computer code and a generic, lowered-loop, 177-FA plant model. The analysis bounds the DBNPS design. The input parameters for the generic B&W analysis and the existing USAR analysis for the DBNPS are compared below.

Parameter	B&W	USAR
Initial Core Power, Wt	2.772	2.772
BOC Moderator Temperature Coefficient, pcm/°F	9.0	1.3
BOC Doppler coefficient, pcm/°F	-1.3	-1.28
Delayed Neutron Fraction, β_{eff}	0.007	0.00689
Control Rod Travel Time to 2/3 Insertion,sec	1.4	1.4
PSV Capacity at 2575 psig, lbm/hr/valve	311,700	336,000
High Flux Trip, percent full power	112.0	112.0
High Flux Trip delay Time, sec	0.4	0.4
High Pressure Trip, psia	2400.0	2400.0
High Pressure Trip Delay Time, sec	0.6	0.6
Number of RCPs Operating	4	4
Total RCS Flow, lbm/hr	139.0E+6	143.4E+6

The initial power level for the startup event is 2.772 Wt, which is 1.0E-9 times the rated thermal power of 2772 MWt. For a subcritical condition (1.0E-9 Wt), the reduced doppler feedback results in the event being terminated by a high flux trip. The high flux trip limits the maximum power increase and results in a lower peak RCS pressure. At 2.772 Wt, the higher fuel temperature allows the doppler feedback to moderate the neutron power increase. This allows the RCS pressure to follow the neutron power and results in a bounding peak pressure. Therefore, the 2.772 Wt power level is conservative for this analysis.

The analysis was performed using beginning of cycle (BOC) kinetics. The smaller (less negative) doppler coefficient, and a larger (more positive) moderator temperature coefficient are conservative for this analysis. The slightly larger negative doppler coefficient (-1.3 pcm/°F) used in the generic analysis remains bounding for the present and future fuel cycles. It is approximately the same as the -1.28 pcm/°F used in the existing USAR analysis. The higher moderator temperature coefficient used in this analysis bounds present and future fuel cycles and is conservative for the calculation of peak RCS pressure. The delayed neutron fraction (βeff) was set at 0.007. The larger value of the delayed neutron fraction is conservative for the startup accident analysis.

The reactor trip is modeled to occur when the neutron power reaches 112% of 2772 MWt or when the primary system pressure reaches 2400 psia (2355 psig + 30 psi error) at the hot leg pressure tap. A trip delay of 0.4 seconds is assumed for the high flux trip. The high pressure trip delay of 0.6 seconds is assumed. These values represent the delay from the time the trip condition is reached to the time the control rods are free to fall. These numbers are equivalent to those used in the USAR analysis and are conservative with respect to the Technical Specification response times.

No credit is taken for actuation of the PORV or pressurizer sprays. The flow rate through the generic PSVs is 311,700 lbm/hr/valve at 2575 psig. This is conservative based on the original USAR value of 336,000 lbm/hr/valve for the DBNPS. Consistent with hot zero power conditions, it is assumed that four reactor coolant pumps are operating with an RCS flowrate of 139.0E+6 lbm/hr and the RCS hot leg pressure and temperature are 2155 psig and 532°F, respectively. Since the analyses were initiated from 2.772 Wt during startup, there is little decay heat. However, the assumed value for decay heat is based on ANS 5.1-1979, Decay Heat Power in Light Water Reactors.

Method of Analysis:

Based on the limiting parameters given above, several cases were run at the +3% PSV lift tolerance and a 180 inches initial pressurizer level to determine the limiting RIR.

In addition to the base case analysis, sensitivity studies were performed to determine the effect of initial pressurizer level and setpoint tolerance on peak RCS pressure. The limiting case was then reanalyzed using one PSV to demonstrate that acceptable results can be obtained with a single PSV operating.

Base Case:

The base case was run with the input parameters shown above. The initial pressurizer level was 180 inches and the PSV lift tolerance was +3%. A spectrum of RIRs was run to determine the worst case RIR with regard to peak RCS pressure. The base case RIR, which produced the peak RCS pressure and thermal power, was found to be 2.05E-4 Δk/k/sec. Higher insertion rates result in the transients being terminated by the high flux trip rather than the RCS high pressure trip and result in a significantly lower peak pressure. The peak thermal power for these analyses was 70.99% with a peak pressure of 2724.7 psia (approximately 2710 psig). Both the thermal power and the peak RCS pressure satisfy the USAR acceptance criteria, resulting in approximately 40 psi margin to the

2750 psig RCS pressure limit. This case forms the basis for the proposed Technical Specification change. The results of the base case analysis are shown below.

RIR ($\Delta k/k/sec$)	Peak RCS Pressure (psia)	Peak Thermal Power (%FP)
1.0E-5	PSV Lift Pressure[1]	21.50
5.0E-5	2684.0	45.60
1.0E-4	2698.9	58.55
1.5E-4	2712.1	65.66
2.05E-4	2724.7	70.99
2.06E-4	PSV Lift Pressure[1]	58.19
4.0E-4	PSV Lift Pressure[1]	43.89
1.0E-3	PSV Lift Pressure[1]	67.14

Note 1: The RCS pressure may not have reached the PSV lift pressure during the computer run, but will eventually with very little overshoot.

PSV Lift Tolerance Sensitivity Study:

The base analysis that produced the highest peak RCS pressure was modified to allow different PSV lift tolerances to determine the PSV lift point that produces a peak RCS pressure below 2750 psig. A +3% lift tolerance was used for the base analysis. Lift tolerances of +4%, +4.5% and +5% were analyzed for the lift tolerance sensitivity study. The thermal power produced by the reactor is unchanged for the higher lift tolerances because the reactor trips at the same time. However, the peak RCS pressure increases as the lift tolerance increases because the primary pressure relief is at a higher pressure.

PSV Lift Tolerance (% over setpoint)	Peak RCS Pressure (psia)	Peak Thermal Power (%FP)
3.0	2724.7	70.99
4.0	2739.8	70.99
4.5	2746.6	70.99
5.0	2753.5	70.99

A +4.5% lift tolerance is considered the maximum acceptable for these initial conditions. The peak RCS pressure is 2746.6 psia (approximately 2732 psig). This tolerance provides approximately 18 psi margin to the 2750 psig RCS pressure limit.

Single Operating PSV Study:

The +4.5% lift tolerance case described in the previous section was modified to allow only one PSV to open. The single operating PSV with a 4.5% lift tolerance produced a peak pressure of 2749.4 psia, 2.8 psi higher than with two operating PSVs.

No. of Operating PSVs	Peak RCS Pressure (psia)	Peak Thermal Power (%FP)
2	2746.6	70.99
1	2749.4	70.99

Initial Pressurizer Level Sensitivity Study:

A sensitivity study was performed to determine the magnitude of the effect of reducing the initial pressurizer level. The base case input parameters (+3% lift tolerance and 2 operating PSVs) were modified to utilize reduced pressurizer levels. The base analysis used an initial indicated pressure level of 180 inches. The initial pressurizer level was reduced to 150 inches and 120 inches. A spectrum of RIRs were run to determine if the reduced pressurizer level changed the limiting RIR. It was found that 2.05E-4 $\Delta k/k/sec$ RIR remains bounding. However, the peak power level increases since the time to reactor trip is increased. The results are shown on the table below.

Initial Pressurizer Level (inches)	Peak RCS Pressure (psia)	Peak Thermal Power (%FP)
180	2724.7	70.99
150	2710.4	71.26
120	2698.7	71.83

The change in the initial level from 180 inches to 150 inches reduced the peak RCS pressure by 14 psi. Reducing the initial level to 120 inches reduced the peak RCS pressure by an additional 12 psi. At 120 inches, the peak RCS pressure is 2698.7 psia (approximately 2684 psig). The margin to 2750 psig is approximately 66 psi.

Summary and Conclusion:

The table below summarizes the results of the analysis:

PSV Lift Tolerance (%)	Initial Pressurizer Level (inches)	Peak RCS Pressure (psig)	Margin to 2750 psig	Peak Thermal Power (%FP)
+3.0	180.0	2710.0	40.0	70.99
+3.0	120.0	2684.0	66.0	71.83
+4.5	180.0	2731.9	18.1	70.99
+4.5[1]	180.0	2734.7	15.3	70.99

Note 1: Single operating PSV

The analysis supports a +4.5% lift tolerance for a single operating PSV with an initial pressurizer level of 180 inches. Additional margin is available because procedural guidance at the DBNPS for hot zero power conditions maintains the pressurizer level at approximately 100 inches. However, for conservatism and to maintain substantial margin in the analysis, the lift tolerance will be limited to +3% for the surveillance testing. This value corresponds to acceptable valve performance as defined in ANSI/ASME OM-1, 1981 Edition. Therefore, there is no adverse effect on nuclear safety.

The proposed administrative changes to TS Bases Section 3/4.4.2 and 3/4.4.3 are associated with the other proposed change and would have no effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. Toledo Edison had reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1 in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because increasing the PSV lift tolerance from +1% to +3% only affects the as-found tolerance of the PSVs. The initial setting tolerance will still be limited to +1%. No hardware modification will be done to the valves which could affect any accident initiators.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because increasing the PSV lift tolerance from +1% to +3% does not affect the radiological releases of any accident previously evaluated in the USAR. This is not a hardware modification and the reactor coolant pressure boundary integrity is unaffected.

2. Not create the possibility of a new kind of accident from any previously evaluated because increasing the PSV lift tolerance from +1% to +3% allows the PSVs to protect the reactor coolant pressure boundary from overpressure transients. This change only affects the allowable lift tolerance. The initial lift setting tolerance is still less than +1%. This change does not modify the valve hardware or alter the operation of the valves. The possibility of the

valves spuriously opening during power operation will not be changed. The valve setpoint with a -3% lift tolerance is well above the normal operating conditions and the RCS high pressure trip setpoint.

3. Not involve a significant reduction in a margin of safety because at the +3% lift tolerance the RCS pressure and the reactor thermal power are still within the USAR acceptance criteria for a control rod withdrawal at low power. This change ensures the Technical Specification lift setpoint tolerances are consistent with the requirements given in the ASME Boiler and Pressure Vessel Code.

CONCLUSION:

On the basis of the above, Toledo Edison has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENT:

Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. B&W Owners Group PSV Lift Tolerance Analysis, 86-1232659-00, May 1994

2. BAW-10043, Overpressure Protection For Babcock & Wilcox Pressurized Water Reactors, May 1972

3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1971 Edition

4. ANSI/ASME OM-1, 1981 Edition

5. DBNPS Updated Safety Analysis Report through Revision 19